NON-PUBLIC?: N
ACCESSION #: 8809290338
LICENSEE EVENT REPORT (LER)

FACILITY NAME: Catawba Nuclear Station, Unit 2 PAGE: 1 OF 6

DOCKET NUMBER: 05000414

TITLE: Manual Reactor Trip During Unit Fast Recovery Caused by Loss of Automatic Feedwater Control To Steam Generator 2C Due to Apparent Control Malfunction
EVENT DATE: 05/28/88 LER #: 88/020/01 REPORT DATE: 09/19/88

OPERATING MODE: POWER LEVEL: 015

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR SECTION
50.73(a)(2)(iv)

LICENSEE CONTACT FOR THIS LER:
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COMPONENT FAILURE DESCRIPTION:
CAUSE: X SYSTEM: SJ COMPONENT: CNV MANUFACTURER: M430
REPORTABLE TO NPRDS: YES
CAUSE: X SYSTEM: SJ COMPONENT: 69 MANUFACTURER: W120
REPORTABLE TO NPRDS: NO

SUPPLEMENTAL REPORT EXPECTED: NO

ABSTRACT: On May 28, 1988, at 2355 hours, a manual Reactor trip was initiated by Control Room personnel to prevent an automatic challenge of the Reactor Protection System on Steam Generator (S/G)) 2C Low-Low level. Operations personnel were performing the Unit Fast Recovery Procedure following a May 27, 1988, trip and were in the process of unisolating the Main Feedwater (CF) control valves. The Control Room Operator had just opened the CF control valve outlet isolation valves and was in the process of opening the inlet isolation valves when S/G 2C narrow range level began to rapidly decrease. Subsequent efforts by Control Room personnel did not stop the level decrease in S/G 2C, and the Reactor was subsequently tripped. The Reactor trip in conjunction with Low-Tavg resulted in a Feedwater Isolation, and the subsequent Low-Low Level signal for S/G 2C resulted in an Auxiliary Feedwater actuation. The Unit was in Mode 1, Power Operation, at approximately 15% power at the time of this incident.

This incident is classified as Event-Cause Code X, Other. It is speculated that debris in the supply air may have prevented the electric to pneumatic (E/P)

converter for 2CF48, the S/G 2C CF Bypass control valve, from operating properly. This incident is also classified a Contributing Event Cause Code X, Other, for two reasons: the apparent inability of 2CF46, S/G 2C CF Main Control valve, to supply flow when position demand was set to 15%, and the decrease in CF flow to S/G 2B at approximately the same time that CF flow to S/G 2C decreased.

END OF ABSTRACT

BACKGROUND:

The Main Feedwater (CF) EIIS:SJ! System supplies feedwater to the Steam Generators (S/Gs) (EIIS:SG) through either the Main or Bypass control valves. When in the manual mode, position demand can be set for these valves from the Control Room. Feedwater flow is directed either through CF or Auxiliary Feedwater (CA) EIIS:BA! nozzles into the S/Gs. Prior to swapping from the bypass to the main control valves and from the CF to CA nozzles, the Unit Fast Recovery procedure requires that the isolation valves on both sides of the CF main control valves be opened. (The CF Main Control valves are known to leak by and are double-isolated on Unit 2.) When the isolation valves are opened, some initial S/G level shrink is expected due to a temperature decrease. This step is performed when the Reactor is at approximately 15% power.

The Auxiliary Feedwater System supplies feedwater to the S/Gs in case there is a loss of CF flow. CA Motor Driven Pumps EIIS:P! A and B will start and supply flow in the event of a Low-Low Level in any of the four S/Gs.

DESCRIPTION OF INCIDENT:

On May 27, 1988, Unit 2 tripped on Low-Low S/G Level (see LER 414/88-19). On May 28, 1988, at 1713 hours, the Unit entered Mode 2, Startup. At 1814 hours, the Unit entered Mode 1, Power Operation, and at 2334 hours, the Control Room Operator (CRO) placed the Main Turbine/Generator on line, per 0P/2/A/6100/05, Unit Fast Recovery procedure. Feedwater was being supplied by CF Pump Turbine (CFPT) EIIS:Pl 2A, through the
F Bypass Control Valves, and into the CA
nozzles. The CRO adjusted CFPT 2A speed per the procedure (to ensure adequate flow to the S/Gs when unisolating the CF Main Control Valves EIIS:FCV), which are known to leak by in the closed position). Between 2339 and 2344 hours, the CRO opened the four CF Main Control Valves outlet isolation valves, per the procedure. The CRO noted no significant changes in narrow range S/G levels, and all four S/G levels were being controlled by the CF Bypass Control Valves in the automatic mode. Between 2344 and 2350 hours, the CRO opened the CF Main Control Valves inlet isolation valves per the procedure, for S/Gs A, B, and C. The CRO

did not open the inlet isolation valve for S/G D because as the inlet isolation valve for S/G 2C reached the intermediate position, he noted a sharp decrease in S/G 2C level. At approximately the same time, the inlet isolation valve for S/G A reached the full open position, unisolating S/G A prior to S/Gs B and C. The CRO verified CFPT 2A differential pressure to be at the proper setpoint (-200 psid). Narrow range levels for S/Gs B and C continued to decrease as the inlet isolation valves for SIGs B and C reached the full open position. The Shift Supervisor (SS) directed the CRO to close the outlet isolation valve for S/G 2C (2CF47), in an effort to isolate the S/G from Main Control Valve leakage. At 2350:46 hours, 2CF47 reached the full closed position. No change in the rate of level decrease for S/G 2C was detected by Control Room personnel. In an effort to feed S/G 2C to increase its level, the SS directed the CRO to throttle open the S/G 2C Main Control Valve (2CF46), and to reopen the outlet isolation valve (2CF47). The CF Bypass Control valves for S/G B (2CF39) and S/G C (2CF48) were at 100% demand with levels still decreasing, so the SS directed the Balance of Plant (BOP) Operator to also throttle open the CF Main Control Valve for S/G B

(2CF37). The BOP Operator then throttled 2CF37 to 5% demand. At 2353:29 hours, the S/G 2C outlet isolation valve (2CF47) reached the full open position, and the CF Main Control Valve to S/G C (2CF46) was being throttled open (by Control Room valve position demand indication) by the CRO to feed S/G 2C. S/G B level began to increase, but S/G C level continued to decrease even though the CRO attempted to throttle 2CF46 to 15% open (by Control Room valve position demand indication). The CRO maintained CFPT 2A discharge pressure as high as possible, to provide as much flow as possible to the S/Gs.

By approximately 2355 hours, S/G 2C narrow range level had decreased to approximately 19%. To prevent an automatic challenge of the Reactor Protection System on S/G Low-Low Level at 17%, the SS directed the CRO to manually trip the Reactor. The manual Reactor trip occurred at 2355:34:173 hours, and the Main Turbine trip on Reactor trip occurred at 2355:34:255 hours. The Low-Low S/G Level signal for S/G 2C was received at 2355:39 hours, resulting in a CA actuation. CA Motor Driven Pumps A and B started and supplied feedwater to the four SIGs, and a Blowdown Isolation occurred. At 2355:47 hours, a Feedwater Isolation signal was received on Reactor trip with Low-Tavg. Plant systems responded normally to the Reactor trip. Operations personnel initiated Work Requests 40553 OPS to investigate and repair 2CF48, 40552 OPS to investigate and repair 2CF46, and 40555 OPS to investigate and repair the electric-to-pneumatic (E/P) converters for all four CF Bypass Control valves.

On May 29, 1988, at 0030 hours, the CRO realigned BB System valves, and at 0138 hours, he secured CA Pump 2A. At 0150 hours, the CRO realigned NM System valves and at 0340 hours, he realigned feedwater valves. At 0440 hours, the CRO secured CA Pump 2B and returned the CA System to standby readiness alignment. During the course of these realignments, IAE personnel were troubleshooting feedwater controls and at 1200 hours, the control card for 2CF48 was replaced. By 1830 hours, the E/P converter for all four CF Bypass Control Valves had been calibrated and replaced. The Unit entered Mode 1, Power Operation, on May 30, 1988, at 0200 hours, under the Unit Fast Recovery Procedure.

CONCLUSION:

This incident is classified as Event Cause Code X, Other. The E/P converters for all four CF Bypass Control Valves were replaced on May 29, 1988. Although the As-Found calibrations of these controllers were found to be acceptable, it is speculated that debris may have entered the E/P converter for 2CF48, causing it to malfunction. This speculation is based on the fact that residue was found in the E/P converter upon inspection. The 7300 Process Control Cabinet Control Cabinet Control Card (Model No. NCB3) for 2CF48, the CF Bypass Control Valve for 9/G 2C, was found to be defective and was replaced. The erratic behavior of this Control Card may have prevented 2CF48 from properly controlling S/G 2C level while in the automatic mode. The inlet isolation valve for S/G 2A was opened (per procedure) prior to opening the inlets for SIGs B and C. Normally this is not a problem, but when coupled with the lack of feedwater control for S/G 2C, it could have resulted in a significant decrease in feedwater flow to S/G C.

The contributing cause to this incident is also classified as Event Cause Code X, Other, due to the inability of the CF Main Control Valve (2CF46) to pass flow as it was throttled to 15% demand position. Work Request 40552 OPS was written to investigate and repair 2CF46. The valve was cycled satisfactorily and the stem was inspected, and no problems were detected. The position demand and actual valve position were compared under Work Request 40642 OPS and were found to correspond satisfactorily. This comparison however was not performed as a result of this incident but as a result of CF Main Control Valve position demand indication problems experienced by Control Room personnel during a subsequent Unit recovery on June 6, 1988. The demand/position indications were found to vary by ten to twenty percent for the other three CF Main Control Valves.

The contributing cause to this incident is also classified as Event Cause Code X, Other, due to the fact that the loss of CF flow to S/G 2B at approximately the same time the S/G 2C CF Bypass Control Valve reduced flow to S/G 2C could not be explained. The level in S/G 2B was successfully recovered however, by throttling open its CF Main Control Valve. ..

It is speculated that during the May 28, 1988, Unit Fast Recovery, the actual position of 2CF46 did not match the Control Room position of 15%. This speculation is based on the following: the Transient Monitor did not show a flow increase for S/G C as it did for S/G B when its Main Control Valve (2CF37) was throttled to 5%; the Alarm Summary Report did not indicate that 2CF46 lifted off its closed limit switch; and the Control Room chart recorder for narrow range S/G 2C level never showed an increase in level. In addition, adequate flow would have been available had 2CF46 opened to 15%, based upon CF flow curves at 15% Reactor power. IAE is developing standing work requests to verify Main Control Valve demand indication against actual valve position.

Following actuation of the CA System, the digital point indication for 2CA4, CA Pump Suction from the Upper Surge Tank, changed status without any operation of the valve. In addition, the digital point indication for CA Pump A discharge pressure did not change status until after two minutes following the pump start.

This incident is reportable to NPRDS. The converter is a model GC-77, manufactured by Moore Products Company. A review of NPRDS indicated six previous failures of this converter.

A review of the Operating Experience Program database shows that there has been one previous incident due to debris in a CF Control Valve E/P converter (see LER 414/87-22). The corrective actions were to implement Standing Work Requests (SWRs) to provide PMs for CF control and bypass control valves during refueling outages, and to implement procedures for calibration of E/P converters. The SWRs for loops A, B and D were performed during the Unit 2 EOC-1 outage, but loop C (which includes 2CF46 and 2CF48) could not be performed due to a lack of time.

A review of NPRDS indicates an approximate 10% failure rate of printed circuit cards. The failure of this control card is not reportable to NPRDS. A search of the Operating Experience Program database shows four Engineered Safeguard Actuations, previous to this incident, which were due to printed circuit card failures in the 7300 Process Control Cabinet: LER 413/86-25, LER 414/86-49, LER 414/87-19 and LER 414/87-27. All failed 7300 cards have been returned to Westinghouse for repair and analysis. Also, a filtered forced air cooling system for the 7300 cabinets was installed on Unit 2 in April, 1988. The corresponding Unit 1 modification (NSM CN-10975) is not yet installed. Failure of 7300 printed circuit cards is a recurring problem.

This incident may have been prevented if a wider narrow range S/G level operating band had been implemented, and if a more responsive CF control system was in use.

CORRECTIVE ACTION:

SUBSEQUENT

(1) Operations personnel initiated Work-Request 40552 OPS to investigate

and rep.air 2CF46, S/G 2C Main Control Valve (2CF46 was cycled and inspected and found to operate properly on May 29, 1988.) The air supply filter for 2CF46 was replaced.

(2) Operations personnel initiated Work Request 40553 OPS to investigate and repair 2CF48, S/G 2C Bypass Control Valve (process control card for 2CF48 was replaced on May 29, 1988). The air supply filter for 2CF48 was replaced.

(3) Operations personnel initiated Work Request 40555 OPS to replace the E/P converters for all four CF Bypass control Valves (replacement and calibration occurred on May 29, 1988).

(4) Performance personnel initiated Work Request 6458 PRF to investigate and repair 2CA4.

PLANNED

(1) Instrumentation and Electrical (IAE) personnel will develop Standing Work Requests (SWRs) to verify Control Room demand position against actual valve position for the CF Main Control Valves. These SWRs are to be performed during each forced outage prior to Unit recovery, and will be placed on the Unit 1 and Unit 2 Trip Lists.

(2) IAE personnel will revise the calibration procedures for the CF Main Control Valves, IP/1-2/A/3010/010, to include more points at which to compare demand and actual position, including the point at which each Main Control Valve begins to open.

(3) The CA System problems identified in the Transient Cycle Mini-Trip Report will be further evaluated and any appropriate corrective action will be negotiated.

(4) NSM #CN20535 will be implemented to provide a wider narrow range span for the S/G level instrumentation.

(5) A task force is investigating possible digital feedwater control systems which may be implemented under Duke Power Station Problem Report (SPR) No. 3423.

SAFETY ANALYSIS :

The transient assessment of this incident showed a normal plant response to the Reactor trip. Reactor power quickly dropped to zero as expected. Pressurizer

pressure did not decrease below 2150 psig or increase above 2250 psig following the trip. Pressurizer level remained between 20% and 40% following the trip. Reactor Coolant (NC) System temperature remained above 550 Deg. F, and Hot Leg temperatures decreased to approximately the same values as the Cold Leg temperatures as expected. NC loop average temperatures decreased quickly following the trip, and Cold Leg temperatures trended along with steam pressure as expected. Main Steam flow fell to zero following the trip. S/G pressure remained greater than 1000 psig and less than 1175 psig following the trip. The post-trip S/G wide range levels for all four S/Gs remained above 56% indication. CA flows were supplied to all four S/Gs and resulted in a sharp increase in narrow range levels, as expected. Subsequently, the CRO adjusted CA flows
o
achieve the desired level recovery rate.

This incident is reportable pursuant to 10 CFR 50.73, Section (a)(2)(iv).

The health and safety of the public were not affected by this incident.

ATTACHMENT 1 TO 8809290338 PAGE 1 OF 1

Duke Power Company Hal B. Tucker
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DUKE POWER

September 19, 1988

Document Control Desk
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Subject: Catawba Nuclear Station, Unit 2
Docket No. 50-414
LER 414/88-20, Revision 1

Gentlemen:

Pursuant to 10 CFR 50.73 Section (a) (1) and (d), attached is Revision 1 to Licensee Event Report 414/88-20 concerning a manual reactor trip during unit fast recovery caused by loss of automatic feedwater control to Steam Generator 2C due to apparent control malfunctions. This event was considered to be of no significance with respect to the health and safety of the public.

Very truly yours,

Hal B. Tucker

LERPGL37/Dl/lcs

Attachment

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*** END OF DOCUMENT ***

ACCESSION #: 8809290344